Entergy

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Fred Dacimo
Site Vice President
Administration

October 23, 2006
Indian Point Unit No. 2
Docket No. 50-247
NL-06-094

Document Control Desk
U.S. Nuclear Regulatory Commission
Mail Stop O-P1-17
Washington, DC 20555-0001

Subject: Licensee Event Report # 2006-003-00, "Manual Reactor Trip Due to a Mismatch Between Reactor Power and Turbine Load Caused by Cycling of Steam Dump Valves After a Power Reduction for Loss of Heater Drain Tank Pumps"

Dear Sir:

The attached Licensee Event Report (LER) 2006-003-00 is the follow-up written report submitted in accordance with 10 CFR 50.73. This event is of the type defined in 10 CFR 50.73(a)(2)(iv)(A) for an event recorded in the Entergy corrective action process as Condition Report CR-IP2-2006-05066.

There are no commitments contained in this letter. Should you or your staff have any questions regarding this matter, please contact Mr. Patric W. Conroy, Manager, Licensing, Indian Point Energy Center at (914) 734-6668.

Sincerely,

Fred R. Dacimo
Site Vice President
Indian Point Energy Center

IE22

Attachment: LER-2006-003-00

cc:

Mr. Samuel J. Collins
Regional Administrator – Region I
U.S. Nuclear Regulatory Commission

U.S. Nuclear Regulatory Commission
Resident Inspector's Office
Resident Inspector Indian Point Unit 2

Mr. Paul Eddy
State of New York Public Service Commission

INPO Record Center

NRC FORM 366 (6-2004) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES: 06/30/2007
LICENSEE EVENT REPORT (LER)	Estimated burden per response to comply with this mandatory collection request: 50 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records and FOIA/Privacy Service Branch (T-5 F52), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

1. FACILITY NAME: INDIAN POINT 2	2. DOCKET NUMBER 05000-247	3. PAGE 1 OF 7

4. TITLE: Manual Reactor Trip Due to a Mismatch Between Reactor Power and Turbine Load Caused by Cycling of Steam Dump Valves After a Power Reduction for Loss of Heater Drain Tank Pumps

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV. NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER 05000
8	23	2006	2006 -	003 -	00	10	23	2006	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE	11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check all that apply)*			
1	☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(i)(C)	☐ 50.73(a)(2)(vii)
	☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
	☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
	☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
10. POWER LEVEL	☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☒ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
68%	☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
	☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
	☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(C)	☐ OTHER
	☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

NAME	TELEPHONE NUMBER *(Include Area Code)*
Eugene O'Donnell, Operations Manager, Unit 2	(914) 736-8202

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX
D	SN	JX	Y006	Y	X	SN	TM	F180	N

14. SUPPLEMENTAL REPORT EXPECTED	15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
☐ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)* ☒ NO				

16. ABSTRACT *(Limit to 1400 spaces, i.e., approximately 15 single-spaced type written lines)*

On August 23, 2006, at 1035 hours, control room (CR) operators initiated a manual reactor trip (RT) due to a mismatch between reactor power and turbine load. Power was initially reduced to 77% for loss of Heater Drain Tank (HDT) pumps then further reduced per Technical Specification 3.2.3 due to axial flux difference outside its required operating limit. CR operators initiated a RT during the further power reduction due to a mismatch between reactor power and turbine load from cyclic operation of the high pressure steam dump (HPSD) valves. All primary safety systems functioned properly. The plant was stabilized in hot standby with decay heat being removed by the main condenser. There was no radiation release. The Emergency Diesel Generators did not start as offsite power remained in-service. The Auxiliary Feedwater system (AFWS) started due to steam generator (SG) low level from shrink effect. Feedwater (FW) isolation and actuation of the AFWS occurred due to a 22 SG high level as a result of overfeed from leakby through the 22 FW low flow bypass valve. The cause of the RT was improper gain settings on the HPSD temperature modules which caused the HPSD's to respond increasingly disproportional to the input signal. The improper settings were attributed to inadequate review during implementation of actions for the Power Uprate Project in 2004. The cause of the loss of the HDT pumps was a failed HDT level controller power supply. Contributing causes included incorrect information in the HPSD module calibration procedure and inadequate procedural guidance. Corrective actions include properly setting the HPSD modules, replacement of the HDT power supply, preparation of a calibration procedure for the HPSD modules, calibration training and procedure revisions. The event had no effect on public health and safety.

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NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* **(17)**

Note: The Energy Industry Identification System Codes are identified within brackets { }

DESCRIPTION OF EVENT

On August 23, 2006, at approximately 1035 hours, with reactor power at approximately 68%, Control Room {NA} (CR) Operators initiated a manual reactor trip (RT) {JC} due to a mismatch between reactor power and turbine load. At approximately 1010 hours, the CR received a Heater Drain Tank (HDT){TK} High/Low level alarm {LA} with HDT level dropping slowly, and HDT Pumps {P} (HDTP) 21 and 22 amps normal and stable. CR operators dispatched a Nuclear Plant Operator (NPO) to investigate the event and the NPO verified to the CR that HDT level was low. Operations entered Alarm Response Procedure (ARP) for Condensate and Boiler Feed (2-ARP-SCF) and directed the NPO to take manual control of the HDT level control system. The NPO reported he was unable to place the HDT level controller (LC-1127) {LC} in manual. At approximately 1014 hours, both HDTPs tripped automatically due to low level. Operators entered Abnormal Operating Procedure 2-AOP-FW-1 and initiated performance of applicable actions. At approximately 1015 hours, operators initiated a load reduction of approximately 250 MWe and commenced a boration of 100 gallons at approximately 10 gpm. During the load reduction, the High Pressure Steam Dump (HPSD) {SB} valves were armed and actuated, and a Main Boiler Feedwater Pump (MBFP) suction cutback was actuated. The Control Bank (CB) D rods {AA} automatically inserted as expected. When the control rods inserted, reactor core axial flux difference (AFD) exceeded the Technical Specification (TS) Limiting Condition of Operation (LCO) 3.2.3 limit and the approaching rod insertion limit alarm was actuated in the CR at approximately 1017 hours. ARP 2-ARP-SAF,"Reactor Coolant System," was entered and operators placed rod control in manual to stop the CB insertion. Operators placed the MBFP master controller to manual to restore suction pressure. Reactor core AFD went out of its target band and its operating envelope due to the control rod insertion. Due to the AFD exceeding the operating limit specified in the COLR, operators entered Technical Specification (TS) LCO 3.2.3 condition C with required action C.1 to reduce power to less than 50% within 30 minutes. At approximately 1030 hours, operators exited AOP-FW-1 and transitioned to Plant Operating Procedure (POP) 2-POP-3.1, and commenced reduction of reactor thermal power to less than 50% per TS 3.2.3. AFD returned to a value within the operating envelope but remained outside the target band. During the load reduction operators experienced unexpected HPSD operation and observed increased cycling of the HPSDs with associated changes in plant parameters. At approximately 1035 hours, with reactor power at approximately 68% and turbine load at approximately 400 MWe, operators observed a significant reduction in turbine load as indicated by the net MWe meter with no operator action. As a result of this condition, CR Operators determined that they did not have adequate control of the power reduction and initiated a manual RT at approximately 1035 hours. All primary safety systems functioned properly. The plant was stabilized in hot standby with decay heat being removed by the main condenser {COND}. There was no radiation release. The Emergency Diesel Generators {EK} did not start as offsite power remained in-service. The Auxiliary Feedwater System (AFWS) {BA} started due to steam generator (SG) {AB} low level from shrink effect and main FW isolated as expected. At approximately 1045 hours, CR Operators observed a rapid rise in 22 SG level and initiated actions to isolate 22 SG FW flow by closing Main FW isolation valve BFD-5-1 and FW low flow bypass valve BFD-90-1. At approximately 1048 hours, a high level in 22 SG due to an apparent leakby of either the main or low flow bypass valve resulted in a FW isolation. The initiation of FW isolation subsequently resulted in actuation of the AFWS in accordance with design. The AFWS was already in operation.

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NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* **(17)**

On August 23, 2006, at 1319 hours, a four hour non-emergency notification was made to the NRC (Log Number 42797) for a reactor trip while critical and included the eight hour non-emergency notification for the actuation of the AFW system. Both notifications were in accordance with 10CFR50.72(b)(3)(iv)(A). The RT event was recorded in the Indian Point Energy Center corrective action program (CAP) as CR-IP2-2006-05066. The HDT controller condition was recorded as CR-IP2-2006-05065 and the HPSD condition recorded in CR-IP2-2006-05081.

The initiator of this event was decreasing HDT level. The discharge of Condensate System {SG} and Heater Drain System (HDS) {SN} supplies water to the suction of the two turbine driven MBFPs which maintains the water level in the four SGs. At 100% power, approximately 27% of the FW flow to the SGs comes from the discharge of the HDS and its loss will result in decreasing SG level as the Feed Water System (FWS) {SJ} will not be able to maintain SG level. Therefore, power reduction is necessary to stabilize SG level. Two HDT pumps take suction from the HDT and pump the water collected in the HDT back to the supply for the MBFPs. HDT level control is maintained by modulating HDT pump discharge valves with level controllers LCV-1127 and 1127A. The HDT discharge valves close on decreasing level and will trip the HDT pumps on HDT low level. Investigation of the HDT level control determined the HDT level controller (LC-1127) power supply (LIC-5003) failed. Upon loss of power to LC-1127, the output signal was suspended in state, maintaining the last output signal. Based on system demand at the time of the failure, the HDT pump discharge valves were maintained at approximately 60% open. The failed level controller condition resulted in slowly decreasing HDT level until it reached the HDT pump low-low level trip setpoint. The level controller (LC-1127) {LC} and power supply (LIC-5003) are manufactured by Yokogawa {Y006}.

The High Pressure Steam Dump System (HPSDS) {SB} cycled during the downpower for this event. The HPSDS consists of twelve valves, instrumentation, and controls that provide a means of discharging main steam directly to the main condensers {SG} to reduce transients on the reactor coolant system {AB} during load rejections or plant trips. Each HPSD is an air-operated valve that can be used to either modulate the steam flow or dump steam by going to its fully open position. The steam dumps and automatic rod control will accommodate a load rejection while reactor power is reduced to a new equilibrium power level. The HPSDS controller mode selection switch during normal operation is the temperature mode. The temperature mode controls the steam dumps after either a load rejection of greater than 10% or a turbine trip. In the temperature mode, the steam dump valves are prevented from operating from the load rejection controller unless the loss of load interlock has been actuated (actuated with either a greater than 10% decrease or 5% per minute ramp decrease as sensed by turbine inlet pressure). A large load rejection (i.e., >10%) requires the actuation of the steam dumps to return the plant to a stable condition. The temperature mode controls the steam dumps to match T(Ref) and RCS T(Avg) after a load rejection of greater then 10% or a ramp rate decrease of 5% per minute rate (loss of load interlock). The HPSD modulating signal during a load rejection is the comparison between RCS T(Avg) and T(Ref). T(Ref) is derived from turbine inlet pressure and compared in the load rejection controller with RCS T (Avg). Since the output of the controller is proportional to the difference between RCS T(Avg) and T(Ref), it is used to control the position of the steam dumps.

NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* **(17)**

When the turbine is in operation, the HPSD Lead/Lag Module TM-412J receives a signal representing average T(Avg) and sends its output to a proportional HPSD controller TC-412J (TM-412M provides a similar control if the turbine is tripped). Lead/Lag Module TM-412J augments the input signal in proportion to a rate of change. For a changing input signal the output signal change will be larger than the input signal change to better respond to transient conditions. The magnitude of the increase in output is determined by the lead gain and lag time settings of the Module. The lead gain is a measure of the magnitude of the impulse output change for a given input change. The lead gain is a dimensionless value (output/input) whereas the lead time has a value of time (seconds). During this event Lead/Lag Module TM-412J was improperly calibrated and resulted in improper cycling of the HPSD valves. The Lead/Lag Module TM-412J is a dynamic compensator manufactured by Foxboro {F180}.

An extent of condition review was performed in which all Lead/Lag components affected by the Power Uprate Project for both Unit 2 and 3 were reviewed to ensure all components were correctly calibrated. All other components were found correctly calibrated.

CAUSE OF EVENT

The direct cause of the RT was Operator initiation of a manual RT as a result of observing a significant mismatch of reactor power to turbine load. The cause of the mismatch was cycling (open/close) of the HPSD valves which resulted in diverting main steam to the main condenser bypassing the turbine thereby reducing turbine load. The HPSD valve cycling also caused fluctuations in pressurizer pressure, pressurizer level, SG levels, and RC average temperature.

The root cause (RC) why the HPSDs did not operate as expected was due to improper gain settings of the HPSD Lead/Lag Temperature Module TM-412J, which caused the HPSDs to respond increasingly disproportional to the input signal. The cause of the improper gain setting was incorrect information in the calibration procedure (ICPM-0060) data sheet for TM-412J, which led to miscalibration (control module TM-412M was also miscalibrated for the same reason). The data sheets specified a lead gain of 10 seconds versus a lead time of 10 seconds that resulted in changing the lead gain value to 10. The incorporation of incorrect information is attributed to inadequate review during implementation of actions for the Power Uprate Project in 2004. Specifically, TM-412J was changed to 10.0 seconds lead time and 5 seconds lag time. Previous settings listed in procedure ICPM-0060 were 8 seconds lead gain and 2.5 seconds lag time. The calibration data sheet of procedure ICPM-0060 specified a lead gain of 10 seconds versus a lead time of 10 seconds. The calibration instructions resulted in changing the lead gain value to 10. The consequence of the calibration error was that the HPSD valves would respond with an amplified response for a given temperature difference and result in improper cycling (open/close) of the HPSD valves.

Contributing causes (CC) include the following: CC1: Procedure ICPM-0060 did not have adequate procedural guidance and contained incorrect information for the calibration of the HPSD Lead/Lag units (TM-412J/TM-412M). A historical review of I&C procedure ICPM-0060 revealed that all previous data sheets for Lead/Lag Temperature Module TM-412J always listed the lead gain setting as 8 seconds although it was always calibrated to a lead gain of 8. All previously performed calibration traces would have led the I&C Technician to calculate the lead gain as opposed to lead time.

CC2: Organization to program interface weakness; There is no procedure for rapid load reduction. Procedural guidance on using a combination of turbine load control, HPSDs, and boration to reduce reactor power would have benefited the operators for reduction of reactor power during short TS action time frames. CC3: Programmatic deficiencies; Insufficient procedure details and inadequate review of impacted procedures/documents. During the implementation of the Improved TS (ITS) the ARPs were reviewed, however, the review failed to identify that a less limiting TS could be applied to 2-ARP-SAF. Placing the Control Bank rods in manual during this event stopped reactor power from continuing to lower during the initial turbine load reduction. However, this action created a large difference between T(Avg) and T(Ref) requiring the HPSDs to open more to maintain RCS temperature. Also, the Abnormal Operating Procedure (AOP) for a loss of FW directs load reductions that would cause HPSD operation. The AOP does not address resetting the loss of load interlock for the HPSDS.

CORRECTIVE ACTIONS

The following corrective actions have been or will be performed under the CAP to address the causes of this event and prevent recurrence.

- The HPSD Lead/Lag Temperature Modules TM-412J and TM-412M were recalibrated to the Westinghouse identified value of 10 seconds lead time and 5 seconds lag time (TM-412J) and 10 seconds lead time and 7 seconds lag time (TM-412M) and the calibration data sheets of procedure ICPM-0060 corrected.
- The HDT Level control power supply (LIC-5003) was replaced.
- A review was performed of Lead/Lag components affected by the unit 2 and unit 3 Power Uprate Project to ensure correct data was incorporated. All other components were found correctly calibrated.
- A calibration procedure will be prepared for the calibration of the Lead/Lag modules. The procedure is scheduled to be complete by December 15, 2006.
- Training of I&C personnel will be provided using a new calibration procedure. Training is scheduled to be completed by March 1, 2007.
- Preparation of a rapid load reduction procedure for Units 2 and 3. Procedure preparation is scheduled to be complete by January 31, 2007.
- The Abnormal Operating Procedures for unit 2 (AOP-FW-1) will be revised to include a provision for resetting the HPSD loss of load interlock. Procedure revision is scheduled to be complete by December 29, 2006.
- Alarm Response Procedure 2-ARP-SAF will be reviewed for revision to allow Control Bank rods to remain in automatic for the conditions; "Approaching Rod Insertion Limit (RIL)," and "RIL" alarm and the requirement to emergency borate at the RIL alarm. Review of the procedure for revision recommendations is scheduled to be complete by October 30, 2006.

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EVENT ANALYSIS

The event is reportable under 10CFR50.73(a)(2)(iv)(A). The licensee shall report any event or condition that resulted in manual or automatic actuation of any of the systems listed under 10CFR50.73(a)(2)(iv)(B). Systems to which the requirements of 10CFR50.73(a)(2)(iv)(A) apply for this event include the Reactor Protection System including reactor trip and AFWS actuation. This event meets the reporting criteria because a manual reactor trip was initiated at 1035 hours, on August 23, 2006, and the AFWS was actuated. The AFWS actuated initially as a result of low SG level from shrink effect from the RT and subsequently due to a high level in the 22 SG as a result of overfeeding of FW due to leakby from the 22 FW low flow bypass valve.

PAST SIMILAR EVENTS

A review of the past three years of Licensee Event Reports (LERs) was conducted for events that involved a RT due to steam dump valve problems or from load swings during a power reduction and no LERs were identified as applicable.

SAFETY SIGNIFICANCE

This event had no effect on the health and safety of the public. There were no actual safety consequences for the event because the event did not initiate accidents and the plant safely shut down as designed. Required primary safety systems performed as designed when the RT was initiated. There were no risk related components out of service at the time of the RT. The AFWS actuation was an expected reaction caused by low SG water level as a result of SG void fraction (shrink), which normally occurs after automatic RT during power operation. The actuation of the AFWS as a result of the FW isolation due to a high level in the 22 SG was in accordance with design. The AFWS was operating at the time FW isolated, therefore there was no impact of this event as the safety function was already being performed.

There were no significant potential safety consequences of this event. The closure of the Heater Drain Tank pump discharge valves resulting in a reduction in FW flow and a subsequent reduction in SG level is a credible alternative condition for which the plant is analyzed. This event was bounded by the analyzed event described in FSAR Section 14.1.9, Loss of Normal Feedwater. A Low-Low water level in any one SG initiates actuation of two motor-driven AFW pumps and a Low-Low water level in any two SGs actuates the steam driven AFW pump. The AFW System has adequate redundancy to provide the minimum required flow assuming a single failure. The analysis of a loss of normal FW shows that following a loss of normal FW, the AFWS is capable of removing the stored and residual heat plus reactor coolant pump heat, thereby preventing either over pressurization of the RCS or loss of water from the reactor coolant system and returning the plant to a safe condition.

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A high level in any SG could result in excessive carryover to the main steam line and damage the main turbine if not isolated. Damage to the main turbine could result in a main turbine protection system trip and a subsequent reactor trip. Water carryover in the steam would not impact the motor driven AFWS which has adequate redundancy to provide the minimum required flow assuming a single failure. FW is isolated to the SGs upon a RT, high SG level, and Safety Injection. A high SG water level greater than 75% of the normal operating span in any SG narrow range level initiates a SG High-High level trip via the SG Water Level Control System which closes the main feed regulating valve and actuates closure of the MBFP discharge valves which trips the MBFPs. A trip of any one of two MBFPs will actuate the start of the AFWS.

The limits on the AFD ensure that the reactor core Heat Flux Hot Channel Factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes. For postulated accidents, the AFD limits ensure that fuel cladding integrity is maintained for these transients. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime. The Reactor Protection System (RPS) is designed to actuate a RT for any anticipated combination of plant conditions, when necessary, to ensure a minimum departure from nucleate boiling (DNB) ratio (DNBR) equal to or greater than the applicable safety analysis limit DNBR. In addition, a manual RT can be initiated by control room operators. The manual RT actuating devices are independent of the automatic trip circuitry. The RPS design is of sufficient redundancy and independence to assure that no single failure or removal from service of any component or channel will result in loss of the protection function. The protection system design is to fail into a safe state or state established as tolerable. Therefore, there are no reasonable or credible alternative conditions that would have resulted in serious consequences.